NON-PUBLIC?: N
ACCESSION #: 9110180039
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PLANT HATCH, UNIT 1 PAGE: 1 OF 9

DOCKET NUMBER: 05000321

TITLE: PERSONNEL ERROR RESULTS IN A MAIN TURBINE TRIP AND A REACTOR SCRAM
EVENT DATE: 09/11/91 LER #: 91-017-00 REPORT DATE: 10/09/91

OTHER FACILITIES INVOLVED: PLANT HATCH, UNIT 2 DOCKET NO: 05000366

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: STEVEN B. TIPPS, MANAGER TELEPHONE: (912) 367-7851
NUCLEAR SAFETY AND COMPLIANCE, HATCH

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 9/11/91, at approximately 0925 CDT, unit 1 was in the Run mode at a power level of 2436 CMWT (100 percent of rated thermal power). At that time, the Reactor Feedwater Pumps and the Main Turbine tripped on a false high reactor water level signal. Subsequently, the reactor scrammed and the Recirculation pumps tripped on Turbine Stop Valve (TSV) closure. Reactor pressure immediately increased to 1102 psig resulting in ten of the eleven Main Steam System Safety Relief Valves opening as designed. Reactor water level decreased to the Low Level 3 and 2 setpoints initiating actuation signals to the Primary Containment Isolation System (PCIS), the Units 1 and 2 Standby Gas Treatment Systems (SGTS), the Reactor Protection System (RPS), the High Pressure Coolant Injection (HPCI) system, and the Reactor Core Isolation Cooling (RCIC) system. Level was restored using the HPCI and RCIC systems. Immediately following the scram, RPS bus 'A' de-energized due to an undervoltage condition being sensed on the bus. This is a possible occurrence when RPS is on alternate supply and the 4160 V buses transfer as designed following a Main Turbine trip. The loss of the RPS bus resulted in initiation signals being transmitted to PCIS, SGTS, and the Main Control Room Environmental Control System pressurization mode logic. The bus was re-energized six minutes following the scram.

The cause of the event was cognitive personnel error on the part of a nonlicensed, contract Health Physics technician. The technician inappropriately leaned a hand-held radiation detector against a scram-sensitive panel. The instrument fell, contacting instrumentation valving and ultimately causing the false high level signal. The technician was counseled. A plant-wide directive will be issued addressing the event.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 9/11/91, at approximately 0925 CDT, Unit 1 was in the Run mode at 2436 CMWT (100 percent of rated thermal power). At that time, the Reactor Feedwater Pumps (RFPs, EIIS Code SJ) and the Main Turbine (EIIS Code TA) tripped on a false high reactor water level signal. The Main Turbine trip initiated closure of the Turbine Stop Valves (TSVs) which further initiated a full Reactor Protection System (RPS, EIIS Code JC) actuation and an end-of-cycle Recirculation pump (EIIS Code AD) trip (EOC-RPT). The actuations/trips functioned as designed: all control rods went full-in and the Recirculation pumps tripped.

The false high reactor water level condition occurred when the high pressure instrument leg (also known as the instrument reference leg) of Reactor Water Level Control (RWLC, EIIS Code JK) system level instruments 1C32-N004A and C depressurized causing the instruments to fail upscale. At the time of the event, a contract, nonlicensed Health Physics technician was performing a radiation survey near instrument rack 1H21-P404B. Upon completing the survey, the technician leaned his hand-held radiation survey instrument, known as a "teletector," against the rack. The instrument fell and struck an instrument rack drain valve, 1B21-N095A-DV3, and/or the valve seal wire. The impact opened the drain valve approximately one quarter of a turn and broke the valve seal wire.

The normally closed drain valve which was struck provides draining capability for an instrument reference leg common to thirteen reactor pressure and reactor water level instruments, including level instruments 1C32-N004A and C which provide inputs to the feedwater control logic and the Main Turbine trip logic. When the drain valve was partially opened, the reference leg depressurized and its associated excess flow check valve (ECFV) 1B21-F049A (a Primary Containment Isolation System (PCIS, EIIS Code JM) valve) closed. Level instruments 1C32-N004A and C failed upscale satisfying the two-out-of-three trip logic for an RFP and a Main Turbine trip on high reactor water level resulting in a trip of the RFPs and the Main Turbine. The purpose of this non-safety protective trip is to prevent water intrusion into the RFP steam driven turbines and into the Main Turbine.

Immediately following the reactor scram, reactor water level began decreasing due to void collapse in the core and the loss of the RFPs. With level instruments 1C32-N004A and C failed upscale, the RFP trip signal was sealed in, precluding restart of the pumps. At twenty seconds following the reactor scram, reactor water level had decreased to the Low Level 3 setpoint, initiating a Group 2 PCIS isolation and a second RPS actuation signal. The Reactor Core Isolating Cooling (RCIC, EIIS Code BN) system was manually started to assist in level control. Level continued to decrease, however, at a slower rate. One

minute and 15 seconds following the reactor scram, reactor water level reached the Low Level 2 setpoint resulting in the initiation of a Group 5 PCIS isolation, a Secondary Containment isolation, and an automatic start of the Unit 1 and 2 Standby Gas Treatment Systems (SGTS, EIIS Code BH). The High Pressure Coolant Injection (HPCI, EIIS Code BJ) system also received an automatic initiation signal; however, it had been manually started just prior to reaching the Low Level 2 setpoint.

With HPCI and RCIC injecting, reactor water level began increasing. The lowest level reached during the transient was 41.9 inches below instrument zero (approximately 122.5 inches above the top of the active fuel). The reactor water level was restored to normal and HPCI was subsequently manually tripped. With RCIC still injecting, reactor water level continued to gradually increase.

At 0934 CDT, the reactor water level had increased to the High Level 8 setpoint at which point RCIC automatically tripped as designed. With no makeup to the vessel other than via the Control Rod Drive system (EIIS Code AA), reactor water level began to slowly decrease. Reactor water level continued to be monitored during scram recovery activities. At approximately 1015 CDT, level was approaching the Low Level 3 setpoint; consequently, a licensed operator manually initiated RCIC. However, before RCIC flow was established to the reactor vessel, the Low Level 3 setpoint was reached initiating a second Group 2 PCIS isolation signal and a second RPS actuation signal on low reactor water level. RCIC was used to restore level to normal and level was maintained within the normal range for the remainder of the event.

Reactor pressure increased as expected immediately following the reactor scram. The Main Turbine Bypass Valves automatically opened to assist in pressure control. Approximately two seconds following the reactor scram, pressure reached its maximum level during the event of approximately 1102 psig. Ten of the eleven Main Steam System Safety Relief Valves (SRVs, EIIS Code SB) opened as designed in response to the pressure transient. SRV 1B31-F013J did not open. Its setpoint plus tolerance is 1111 psig and was not reached during the transient; therefore, the valve was not required to open. As reactor pressure decreased, the SRVs sequenced closed as controlled by the SRV Low Low Set (LLS) relief logic. At 42 seconds following the reactor scram, pressure had decreased to approximately 850 psig, the bypass valves had closed, and the SRVs had closed. The pressure then slowly increased to approximately 945 psig and stabilized, being controlled by automatic operation of the Bypass Valves.

Prior to this event, RPS bus 'A' had been placed on alternate supply so that the 'A' RPS motor-generator (MG) set could be removed from service for preventive maintenance. As expected following the turbine trip, the 4160 V non-essential buses transferred from the unit auxiliary transformers to the startup transformers. Also, as expected, the transfer caused a voltage perturbation on the electrical distribution system. The RPS bus undervoltage instrumentation sensed the perturbation and initiated a trip of the 'A' RPS bus protection breakers, de-energizing the bus.

The loss of power on RPS bus 'A' resulted in a loss of power to the Division I Engineered Safety Feature (ESF) actuation logic powered from the bus. Upon the loss of power, the ESF logic, being of a fail-safe design, initiated a Secondary Containment isolation, an automatic start of Unit 1 and 2 SGTS, an isolation of the inboard PCIS Groups 2 and 5 valves, and an automatic transfer of the Main Control Room Environmental

Control (MCREC, EIIS Code VI) system to the pressurization mode. All systems operated as designed.

Following loss of power to the RPS bus, a licensed operator was dispatched to re-energize the bus. Approximately six minutes following the loss of the bus, the bus protection breakers were closed, restoring power to the bus.

Depressurization of the instrument reference leg caused the following ESF instrumentation to behave in the following ways:

o 1B21-N095A, Reactor Water High Level 8 RCIC trip: The instrument failed upscale initiating a trip in its associated channel. The trip logic design is a two-out-of-two scheme. Consequently, a second trip signal is required for the trip system to actuate. This second trip signal originates from physically independent instrumentation and, thus, was not affected by the EFCV closure. Consequently, the instrument failing upscale did not cause a premature trip of the RCIC system.

o 1B21-N093B, Reactor Water High Level 8 HPCI trip: The instrument failed upscale initiating a trip in its associated channel. The trip logic design is a two-out-of-two scheme. Consequently, a second trip signal is required for the trip system to actuate. As with the RCIC system, this second trip signal originates from physically independent instrumentation and, thus, was not affected by the EFCV closure. Consequently, the instrument failing upscale did not cause a premature trip of the HPCI system.

o 1B21-N080A, B, Reactor Water Low Level 3 RPS actuation: These two instruments failed upscale preventing the generation of an RPS actuation signal on an actual Low Level 3 condition in two of the trip system's four instrument channels. The trip system's logic design is such that the remaining two operable instruments and channels, being unaffected by the closure of the EFCV, were sufficient to initiate an RPS actuation on a Low Level 3 condition. Consequently, RPS actuated per design upon reaching the Low Level 3 setpoint during the event.

o 1B21-N080A, B, Reactor Water Low Level 3 PCIS Group 2 isolation: These two instruments failed upscale preventing the generation of an isolation signal on a Low Level 3 condition to the inboard Group 2 PCIS valves. The inboard Group 2 PCIS valves had cycled closed due to RPS bus 'A' de-energizing prior to the Low Level 3 setpoint being reached in the event. Also, the redundant and independent Group 2 PCIS outboard valves were unaffected by the failed instruments and closed as required during the event.

o 1B21-N095A, Reactor Water Low Level 3 Automatic Depressurization System (ADS) permissive: The instrument failed upscale preventing generation of a permissive signal to ADS on a Low Level 3 condition. This condition rendered inoperable one of two ADS initiation logic trains. The other redundant and independent initiation logic train was unaffected and, had it been called on to do so, would have initiated ADS as designed.

o 1B21-N085A, Reactor Water Low Level 0 Residual Heat Removal (RHR) System Containment Spray permissive: This instrument failed upscale erroneously indicating that the reactor water level was sufficient to allow cooling water from the Low Pressure Coolant Injection (LPCI, EIIS Code BO) mode of RHR to be diverted for containment spray purposes. However, neither the LPCI mode nor the Containment Spray mode of RHR were required to operate during the event. If Containment Spray (a manually actuated mode of RHR) had been required to operate during the event, sufficient guidance exists in the Emergency Operating Procedure (EOP) flowcharts and procedure 34SO-E11-010-1S, "Residual Heat Removal System," to ensure that it would not have been initiated prematurely.

Limiting Condition of Operation (LCO) 1-91-483 was initiated to address the Technical Specifications operating limitations associated with the above noted instrumentation being inoperable. The LCO, per Unit 1 Technical Specifications sections 3.1, 3.2, 3.5 and 3.7.B, required the unit to be in Cold Shutdown within 24 hours of the instruments becoming inoperable. At approximately 1319 CDT, approximately 4 hours after the initiating event, drain valve 1B21-N095A-DV3 was found to be partially open. The drain valve was subsequently closed, the reference leg was repressurized, and the EFCV was opened. With the reference leg repressurized, the above noted instruments were operable and the LCO was subsequently terminated.

It was noted during the event that the Unit 1 SGTS suction dampers automatically closed following reset of the Group 2 and 5 PCIS signal. This is in conflict with the requirements of IE Bulletin 80-06 and was reported in LER 50-321/91-14, dated 9/9/91.

CAUSE OF EVENT

The cause of the event was cognitive personnel error on the part of nonlicensed personnel. Specifically, a contract Health Physics technician inappropriately leaned a hand-held radiation detector against an instrument rack containing 'scram-sensitive' instrumentation. The detector, not being properly supported, fell and struck a drain valve and/or its seal wire located on the instrument rack. The impact of the fall opened the normally closed drain valve approximately one quarter of a turn and broke the seal wire. The drain valve, 1B21-N095A-DV3, taps off of the high pressure instrument leg common to thirteen reactor water level and reactor pressure instruments including two instruments, 1C32-N004A and C, which provide input to the RWLC system logic and the Main

Turbine trip logic. When the drain valve was opened, the instrument leg depressurized, causing instruments 1C32-N004A and C to peg upscale and to generate a high reactor water level trip to the RFPs and the Main Turbine. A reactor scram subsequently occurred on TSV closure.

REPORTABILITY ANALYSIS AND-SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) because the event involved the automatic actuation of RPS and other ESF systems. Specifically, a full RPS actuation occurred on TSV closure and RPS actuation signals were generated when reactor water level reached the Low Level 3 setpoint. Automatic initiations of Group 2 and 5 PCIS, Unit 1 and 2 SGTS, Secondary Containment isolation, and the pressurization mode of the MCREC system occurred on a loss of RPS bus 'A'. Automatic actuation of the Group 2 PCIS occurred when reactor water level reached the Low Level 3 setpoint. Also, automatic actuations of Group 5 PCIS, Secondary Containment isolation, Unit 1 and 2 SGTS, and the HPCI system occurred when the reactor water level reached the Low Level 2 setpoint.

In the event, a full RPS actuation occurred on TSV closure. The purpose of RPS is to initiate a reactor scram when specified plant parameters are exceeded to ensure that the radioactive materials barriers, such as fuel cladding and the pressure system boundary, are maintained and to mitigate the consequences of transients and accidents. Closure of the Turbine Stop Valves, such as occurs on a Main Turbine trip, can result in the addition of positive reactivity to the core as the resultant reactor pressure increase collapses coolant voids. Therefore, Turbine Stop Valve closure initiates a scram in anticipation of high neutron flux or high reactor pressure conditions, which also initiate scram signals. The high pressure scram, in conjunction with the pressure relief system, and the high neutron flux scram are adequate to preclude failure of the radioactive materials barriers; however, the Turbine Stop Valve closure scram provides additional margin. RPS performed as designed in the event. A false high reactor water level signal resulted in a Main Turbine trip being initiated and the TSVs closing. As a result of the TSVs closing, RPS actuated, effecting a reactor scram.

Turbine Stop Valve closure also initiates a trip of the EOC-RPT logic whenever the Main Turbine first stage pressure is above that which corresponds to 30 percent of rated thermal power. This logic functions to trip the Recirculation pumps in response to a Main Turbine trip. By tripping the Recirculation pumps early in the event, the severity of the Main Turbine trip is reduced. The rapid reduction in core flow reduces coolant void collapse during pressurization events resulting in less positive reactivity being added to the core. As with the scram on Turbine Stop Valve closure, the EOC-RPT provides a satisfactory margin for core thermal-hydraulic safety limits. In the event, the Recirculation pumps tripped on TSV closure as designed.

The SRVs and Low Low Set (LLS) relief logic, the HPCI system, and the RCIC system operated to control reactor pressure and/or reactor water level during the event. The purpose of the SRVs is to prevent overpressurization of the reactor and steam supply system in order to preclude failure of the nuclear system process barrier. The purpose of the LLS relief logic system is to mitigate the effects of postulated thrust loads on the SRV discharge lines and the effects of postulated high-frequency pressure loads on the suppression pool (EIIS Code BT) shell caused by subsequent actuations of the SRVs during a small or intermediate break LOCA. This is accomplished by extending the time between SRV subsequent actuations to allow the SRV discharge line water leg to return to its original level.

During the event, reactor pressure increased as expected upon closure of the TSVs. The peak pressure reached during the transient was 1102 psig which is considerably less than the reactor pressure safety limit of 1325 psig. Ten of the eleven SRVs opened in response to the transient. The pressure setpoint plus tolerance of SRV 1B21-F013J, the SRV that did not open, is 1111 psig. Since this pressure was not reached in the transient, it was not required to open. With its reactor pressure setpoint permissive reached and an SRV open, the LLS relief logic was armed, As reactor pressure decreased, LLS relief logic controlled the closing sequence of the SRVs as designed. By the time reactor pressure decreased to approximately 850 psig, all ten SRVs had closed. Reactor pressure remained stable for the remainder of the event being controlled by automatic operation of the Bypass Valves.

The HPCI system functions to provide adequate reactor core cooling to limit the fuel-clad temperature in the event of a small break in the nuclear steam system which does not result in a rapid depressurization of the reactor vessel. The HPCI system automatically starts on a reactor water Low Level 2 signal or a high Drywell pressure signal. The RCIC syst
m functions to provide adequate core cooling when the reactor is shutdown and the feedwater makeup to the vessel is lost. The RCIC system automatically starts on a reactor water Low Level 2 signal.

Immediately following the reactor scram, the reactor water level began decreasing as expected due to void collapse and the loss of feedwater makeup to the vessel. The false reactor water high level signals from instruments 1C32-N004A and C were sealed in due to the instrument reference leg being depressurized. Consequently, the RFPs could not be restarted to assist in reactor water level control. Approximately 35 seconds after the reactor scram, RCIC was manually started to assist in controlling reactor water level. Water level continued to decrease, however, at a much slower rate. One minute and 15 seconds following the reactor scram, the Reactor Water Low Level 2 setpoint was reached at which time an automatic start signal for the HPCI and RCIC systems was initiated. The RCIC system was already operating at the time and the HPCI system had been manually initiated just prior to reaching the level setpoint. Before HPCI system flow was established, the lowest reactor water level experienced during the event was reached, 41.9 inches below instrument zero (122.5 inches above the top of the active fuel). The HPCI and RCIC systems operated in tandem to restore level.

The Primary Containment Isolation System provides timely protection against the onset and consequences of events involving the potential release of radioactive materials from the fuel and nuclear system process barriers by isolating appropriate lines which penetrate the primary containment. Group 2 and Group 5 Primary Containment Isolation System isolations which are initiated by reactor water Low Level 3 and 2 signals, respectively, prevent the escape of radioactive materials from the primary containment through process lines which may have been breached. Additionally, isolation of these process lines conserves reactor coolant inventory if a breach of one of these lines caused the low water level condition. In the event, PCIS operated as designed upon receipt of the appropriate isolation signal.

Reactor vessel instrumentation provides reliable monitoring of critical reactor vessel parameters and provides the appropriate trips/initiating signals upon sensed parameters exceeding a design setpoint. The instrumentation is designed such that a single failure would not prevent the initiation of a safety system function. In this event, an instrument high pressure leg was depressurized resulting in thirteen reactor pressure and reactor water level instruments failing either upscale or downscale, depending on the parameter being monitored.

As previously addressed in the Description of Event, the instrumentation and actuation logic associated with the failed instrumentation is designed such that, with the instrument leg depressurized, the ESF systems received initiation signals as designed when the appropriate parameter setpoints were reached during the event. Specifically, PCIS initiation signals were generated as designed when reactor water level reached the Low Level 3 and 2 setpoints. RPS initiation signals were generated as designed when reactor water level reached the Low Level 3 setpoint. Neither the HPCI nor the RCIC systems prematurely tripped when half of the level instruments associated with their high level trip function failed upscale. Conversely, when the High Level 8 setpoint was actually reached, trip signals to the HPCI and RCIC systems were initiated as designed. Also, ADS, had it been required to operate during the event, would have operated as designed since it received the required Low Level 3 permissive via an independent, redundant trip system in spite of the instrument leg failure. Containment Spray (a manually actuated mode of RHR) was also not required to operate during the event. The instrument providing a 2/3 core height permissive for placing one of the two RHR loops in the Containment Spray mode failed upscale when the instrument leg depressurized causing the permissive to seal in regardless of the water level. As stated previously, the EOP flowcharts in conjunction with procedure 34SO-E11-010-1S ensure that the RHR system is placed in the Containment Spray mode only after the appropriate plant limits are met.

The 125 V RPS bus provides power to ESF actuation logic as well as to RPS. The bus is normally powered from an MG set. The MG set, which itself is powered from an essential 600 V bus, has an inherent flywheel effect which aids in maintaining close tolerances on voltage and frequency. With the RPS bus fed from its alternate power supply (an essential cabinet) the MG set is bypassed and, thus, the bus voltage and frequency are more likely to be affected by minor perturbations on the station service transformers.

Immediately following the Main Turbine trip, RPS bus 'A' de-energized when its bus protection breakers tripped. The breakers tripped as designed upon sensing a voltage perturbation on the bus. The perturbation resulted from the 4160 V bus transfers which normally follow a Main Turbine trip. As designed, upon a loss of power on RPS bus 'A', the ESF logic initiated a Secondary Containment isolation, an automatic start of the Unit 1 and 2 SGTS, an isolation of the inboard PCIS Groups 2 and 5 valves, and an automatic transfer of the MCREC system to the pressurization mode. All systems actuated as designed.

Based on the above information, this event had no adverse impact on nuclear safety. This assessment is applicable to all power levels.

CORRECTIVE ACTIONS

The contract Health Physics technician was counseled regarding the consequences of his actions.

A plant-wide directive has been issued by the General Manager addressing this event and emphasizing the need to exercise extreme caution when working around plant equipment.

ADDITIONAL INFORMATION

No systems other than those previously mentioned in this report were affected by this event.

No failed components were involved in this event.

Previous similar events in the last two years in which the reactor scrammed as a direct result of personnel error were addressed in the following reports:

50-321/91-17, dated 3/27/91
50-366/91-05, dated 3/15/91

Corrective actions for these events would not have prevented this event because they involved different personnel and modes of error unique to the activities that resulted in the scram.

ATTACHMENT 1 TO 9110180039 PAGE 1 OF 2

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October 9, 1991

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNIT 1
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
LICENSEE EVENT REPORT
PERSONNEL ERROR RESULTS IN
A MAIN TURBINE TRIP AND A REACTOR SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a personnel error which resulted in a main turbine trip and a reactor scram. This event occurred at Plant Hatch - Unit 1.

Sincerely,

J. T. Beckham, Jr.

MCM/et

Enclosure: LER 50-321/1991-017

cc: (See next page.)

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

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*** END OF DOCUMENT ***